August 29, 2014

Dr. Leah Jamieson, Dean
College of Engineering
Purdue University
400 Central Dr.
West Lafayette, IN 47907-2017

SUBJECT: PURDUE UNIVERSITY - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PURDUE UNIVERSITY REACTOR LICENSE RENEWAL APPLICATION (TAC NO. ME1594)

Dear Dr. Jamieson:

The U.S. Nuclear Regulatory Commission is continuing its review of the Purdue University application for the renewal of Facility Operating License No. R-87 by letter dated July 7, 2008 (Agencywide Documents Access and Management System Accession No. ML083040443), as supplemented, for the Purdue University Research Reactor (PUR-1).

During our review, questions have arisen for which we require additional information and clarification. Provide responses to the enclosed request for additional information no later than 30 days from the receipt of this letter.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.30(b), your response must be executed in a signed original document under oath or affirmation. You must submit your response in accordance with 10 CFR 50.4, "Written communications." Information included in your response that is considered sensitive or proprietary, that you seek to have withheld from the public, must be marked in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Any information related to security should be submitted in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance requirements." Following receipt of the additional information, we will continue our evaluation of your license renewal request.

If you have any questions regarding this review, please contact me at 301-415-3398 or by electronic mail at Cindy.Montgomery@nrc.gov.

Sincerely,

/RA/

Cindy K. Montgomery, Project Manager
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-182

Enclosure:
Request for Additional Information

cc: See next page

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ADAMS Accession No.: ML14115A221 **NRR-088**

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202 Nuclear Sciences Center
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REQUEST FOR ADDITIONAL INFORMATION

FOR THE LICENSE RENEWAL FOR

PURDUE UNIVERSITY RESEARCH REACTOR

LICENSE NO. R-87

DOCKET NO. 50-182

The U.S. Nuclear Regulatory Commission (NRC) is continuing its review of the Purdue University application for the renewal of Facility Operating License No. R-87 by letter dated July 7, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML083040443), as supplemented, for the Purdue University Research Reactor (PUR-1).

During our review, questions have arisen for which we require additional information and clarification. Provide responses to the enclosed request for additional information (RAI) no later than 30 days from the receipt of this letter.

The purpose of these questions is to assist the NRC staff in determining if the renewal application for PUR-1 meets the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation," and Part 50, "Domestic Licensing of Production and Utilization Facilities." The following RAIs are, in part, based on a comparison of the PUR-1 license renewal application with NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," dated February 1996 and with the American National Standard Institute/American Nuclear Society (ANSI/ANS)-15.1-2007, "The Development of Technical Specifications for Research Reactors." NUREG-1537, Part 1, Appendix 14.1 suggests that the format and content of the proposed Technical Specifications (TSs) follow the recommendations of ANSI/ANS-15-2007.

In July 2011, the NRC issued three sets of RAIs: (1) questions that had a 30-day requested response time, (2) questions that may have required outside resources for assistance in completion and had a 60-day requested response time, and (3) complex and computational questions that had a 90-day requested response time. The three groups of questions resulted in a complete set of consecutively numbered questions 1-99.

RAIs 1-2 are based on a review of your RAI responses to the NRC's letter dated July 6, 2011 (ADAMS Accession No. ML101460429), that required a "30-day" response.

1. RAI 18 in NRC letter dated July 6, 2011, stated:

> TS 4.3: TS 4.3(c) should reference the minimum 13 foot depth as specified in the LCO [limiting condition for operation] (TS 3.3(c)) for primary coolant and provided in the bases for TS 4.3. Please update the TS to include the numerical minimum depth and surveillance interval for this surveillance or justify why an alternative measure related to the height of the skimmer trough in TS 4.3(c) is more appropriate for specifying the minimum performance level of TS 3.3(c). Additionally,

> prescribe the frequency, scope and minimum water level of this surveillance when the reactor is secured or shutdown or justify why a minimum level is not required.

The response to RAI 18 by letter dated January 4, 2012 (ADAMS Accession No. ML12006A193), proposed to modify TS 4.3(c) to specify that reactor pool water will be at a height of the 13 foot over the top of the core whenever the reactor is operated. However, the response to RAI 17 by letter dated January 30, 2012 (ADAMS Accession No. ML12031A223), proposed to modify TS 4.3(c) to specify the reactor pool water will be at or above the height of the skimmer. Clarify TS 4.3(c), ensuring that RAI 18 is answered clearly and completely. Include a basis for any TS changes proposed.

2. RAI 29 in NRC letter dated July 6, 2011, stated:

> TS 6.1.11, TS 6.1.14: ANSI/ANS-15.1-2007, Section 6.1.3(3) provides guidance for events requiring the presence at the facility of the senior reactor operator. Please update PUR-1 TS 6.1.11 and 6.1.14 for compliance with the requirements in ANSI/ANS-15.1-2007, Section 6.1.3(3) and 10 CFR 50.54(m)(1) or provide an explanation describing your reason(s) for not incorporating the changes.

The response to RAI 29 by letter dated January 30, 2012, proposed a modification of TSs 6.1.11 and 6.1.14 that does not meet the requirements of 10 CFR 50.54(m)(1) which state:

> A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

The response proposed a change to TS 6.1.2 (Staffing), Item (3), stating:

(3) Events requiring the presence at the facility of an senior reactor operator [SRO] are:

(a) Initial startup and approach to power following a core change. The presence of an SRO at the reactor facility is unnecessary for the initial daily start-up, provided the core remains unchanged from the previous run;

(b) All fuel or control-rod relocations within the core region;

(c) Recovery from an unplanned or unscheduled shutdown except in instances which result in the following

(i) A verified electrical power failure …;

(ii) Accidental manipulation of equipment in a manner which does not affect the safety of the reactor;

(iii) A verified practice of the evacuation of the building initiated by persons exclusive of reactor operations personnel.

Provide an explanation on how the proposed changes (a) and (c) are in compliance with the requirements of 10 CFR 50.54(m)(1).

RAIs 3-4 are based on a review of your RAI response to NRC's letter dated July 8, 2011 (ADAMS Accession No. ML103400115), that required a "60-day" response.

3. RAI 44 in NRC letter dated July 8, 2011, stated:

> In your RAI response concerning decommissioning cost, dated June 4, 2010, you reference an "approved cost estimate for decommissioning under Purdue University's Radioactive Materials License" as the basis for the provided cost estimate. Please describe and explain the relationship between the decommissioning cost estimate for PUR-1 and cost estimate for decommissioning under Purdue University's Radioactive Materials License in determining the cost estimate for decommissioning PUR-1.

Provide a response to RAI 44, since we have not yet received one.

4. RAI 45 in NRC letter dated July 8, 2011, stated:

> Pursuant to 10 CFR 55.59(a)(2), each licensee shall: "Pass a comprehensive requalification written examination and an annual operating test." In your Requalification Plan, Section B you state that "[c]ompletion of the biennial requalification program will consist of a written examination and a demonstration of operator proficiency in reactor operation."
>
> A. Explain how the facility ensures that operator proficiency examinations are performed annually during the biennial requalification cycle in compliance with 10 CFR 55.59(a)(2) or update your plan accordingly.
>
> B. As required by 10 CFR 55.53(h), licensees are required to complete a requalification program as described by 10 CFR 55.59. The regulation in 10 CFR 55.59(a) states that each license shall:

(1) Successfully complete a requalification program developed by the facility licensee that has been approved by the Commission. This program shall be conducted for a continuous period not to exceed 24 months in duration.

(2) Pass a comprehensive requalification written examination and an annual operating test.

Section F of the PUR-1 Requalification Plan states:

> During intervals when the licensed operations crew consists only of senior operators who are instructors for topics in part a.1.b., the requalification program will be modified to exempt those senior operators from parts A and B.1. Parts B.2, C, D, and E will remain in effect.
>
> When the licensed operations crew increases to include those who do not instruct in the program, the program will revert to its initial content. Operators may place a statement into the file stating that they have done a literature review and/or instructed the topics in Section A and B.1 in lieu of meetings and exams.
>
> During intervals when the licensed operations crew consists of only one senior operator this operator will be exempt from parts A and B, part C would be documented in the console logbook and as stated in C.3, parts D and E will remain in effect.
>
> In any of the requalification activities, exclusive of operations, additional methods may be used to accomplish the training requirement. These may include mail, electronic classroom or other methods may be used for training, meetings, testing or other required communication(s).

The response to RAI 45 by letter dated January 31, 2012 (ADAMS Accession No. ML14234A109, redacted version) indicated that the PUR-1 facility previously had an exemption and intends to request one again. Either (a) explain how this section, in its current form, meets the requirements of 10 CFR 55.33, “Disposition of an initial application” and 10 CFR 55.59; (b) delete this section of the requalification plan; or (c) submit an exemption for these requirements in accordance with 10 CFR 55.11, “Specific exemptions.”

RAIs 5-7 are based on a review of your RAI response to NRC's letter dated July 14, 2011, that required a "90-day" response (ADAMS Accession No. ML103400250).

5. RAI 63 in NRC letter dated July 14, 2011, stated:

> Major inconsistencies are noted throughout the SAR [safety analysis report] related to calculation assumptions for initial and requested maximum licensed power under the PUR-1 license renewal. For example, SAR Section 13.2.2, p. 13-11, references current licensed power of 1 kW [kilowatt] for a reactivity insertion with scram. Please clarify the desired maximum licensed power level requested and ensure this power level, including any uncertainty in reactor power, is consistently applied in the safety analyses for the license. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested maximum licensed power level.

Additional clarification to RAI 63 is needed. Provide responses to the questions below:

(a) Provide an answer to RAI 63 or indicate if the answer to RAI 63 is provided in the responses to RAIs 62 and 65 in your letter dated January 31, 2012.

(b) Table 4-21 in the revised PUR-1 SAR, Section 4.6 indicates that for 1 kW and 12 kW, the maximum fuel temperatures for the limiting fuel plate (31.92 and 39.1 degrees Celsius (C)) are lower than the maximum clad temperatures (43.42 and 43.4 degrees C). Discuss the physical phenomena that would result in these temperature values.

6. RAI 70 in NRC letter dated July 14, 2011, stated:

> NUREG-1537, Section 11 provides guidance for radiation protection provisions at the facility. In Section 4.4 of the SAR, it is stated that the radiation level above the reactor pool surface is about 1 mrem [millirem]/hr and that the radiation level along the outside lateral surface of the concrete biological shield is about 0.1 mrem/hr, when the core is operating at 1 kW. Please provide an updated evaluation of a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential radiation levels and the potential radiation effects on facility staff. As part of evaluation, please indicate if the radiation levels bound those that would be encountered during fuel handling and maintenance operations. Additionally, include an evaluation of the safety analysis for potential dose to the facility staff and members of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.).

The response to RAI 70 by letter dated January 31, 2012, did not address the expected radiation dose levels at the requested increased licensed power level of 12 kW. The updated safety analysis should indicate bounding dose levels for facility staff and members of the public who may be located in nearby or adjacent, accessible public

areas during the maximum operational power level for an extended period (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.) to demonstrate compliance with 10 CFR Part 20. Include all analyses, assumptions, and conclusions and indicate if the radiation levels bound those that would be encountered during fuel handling and maintenance operations.

7. RAI 71 in NRC letter dated July 14, 2011 stated:

> The requirements of 10 CFR 20.1101 states that each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities, in order to limit the total effective dose equivalent to facility workers (annual occupational dose less than 5 rem [roentgen equivalent man]) and the total effective dose equivalent to individual members of the public (annual public dose less than 100 mrem). Please provide … a safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential estimate of the total annual production of argon-41 from PUR-1 normal operations. In addition, please evaluate and discuss the potential maximum dose to a facility worker and to a member of the public (i.e., classrooms, hallways, adjacent rooms, nearest dormitories, offices, etc.) due to this bounding yearly production and release of argon-41 from the facility.

Your response to RAI 71 by letter dated April 10, 2013 (ADAMS Accession No. ML13101A044), did not provide an analysis to determine the maximum effective dose to the maximally exposed member of the public for the total annual production of argon-41 from PUR-1 maximum licensed power operations. Provide a bounding safety analysis (with assumptions and conclusions) for the member of the public residing outside the facility perimeters.

8. In letter dated April 10, 2013, Purdue University provided an updated Section 13 of the SAR that included responses to RAIs 74-76, 80, 83, 85-91, and 93-95. State whether the updated SAR Section 13 also intended to provide answers to RAIs 79, 81, 82, 84, and 99 or provide answers:

(a) RAI 79 in NRC letter dated July 14, 2011, stated:

> NUREG-1537, Section 13 provides guidance to identify the limiting event for each accident group and to perform quantitative analysis for that event. Please identify the categories of PUR-1 experiments that are performed and provide an evaluation of a safety analysis using the guidance of NUREG-1537, Section 13.1.6 for potential experiment malfunctions and their consequences.

(b) RAI 81 in NRC letter dated July 14, 2011, stated:

> NUREG-1537, Part 1, Section 13.1.1 provides guidance to identify Maximum Hypothetical Accidents (MHA) for non-power reactors. The MHA is to be selected so potential consequences of the postulated MHA scenario exceed and bound all credible accidents. NUREG-1537, Part 2, Chapter 13, p. 13-5 suggests that for a low-powered MTR fueled reactor, the MHA may be one of the following two events: cladding is stripped from one face of a fuel plate while suspended in air, or a fueled experiment fails in air. SAR, Section 13.1 states that “the failure of a fueled experiment is designated as the maximum hypothetical accident of the PUR-I.” Please provide … a safety analysis of an MHA that considers the failure of one fuel plate in air would have lower consequences than the failure of a fueled experiment by justifying the MHA accident involving the fueled experiment capsule is more bounding than the failure of a fuel plate.

(c) RAI 82 in NRC letter dated July 14, 2011, stated:

> NUREG-1537, Part 1, Section 13.1.1 provides guidance in identifying an acceptable MHA for non-power reactors. The PUR-1 MHA accident analysis for “Failure of a Fueled Experiment” is stated to be based upon a 1 W power deposition in the fueled experiment as consequence of the reactor operating at 1 kW. Please provide … a safety analysis that provides the details of the energy deposition determination in the fueled sample with the reactor operating at the maximum requested licensed reactor power including the power level measurement uncertainty of 50% stated in SAR, Section 13.1.2.

(d) RAI 84 in NRC letter dated July 14, 2011, stated:

> NUREG-1537 states that the format and content of the TS follow ANSI/ANS 15.1. ANSI/ANS-15.1-2007, Section 3.8.2 provides guidance for double encapsulation of experiments involving fissionable, explosive, reactive, or corrosive materials. Please Provide … a safety analysis for the MHA experiment of 1.1 g of U-235 with single encapsulation is consistent with the guidance provided in ANSI/ANS-15.1-2007, Section 3.8.2.

(e) RAI 99 in NRC letter dated July 14, 2011, stated:

> SAR, Section 13.2.1, page 13-9 states “*This experiment corresponds to the irradiation of 1.1 gm of U-235 in the mid-plane of the isotope irradiation tube located in position F6.”* Please provide …a safety analysis that establishes the basis of 1.1 gm of U-235 for failure of a fueled experiment.

The following RAIs are based on the guidance in NUREG-1537.

9. In a letter dated April 10, 2013, Purdue University provided an update to the SAR, Section 13 containing an analysis for the designated MHA based on the failure of a fuel plate and not the malfunction of a fueled experiment. Provide an analysis determining whether the consequences of the failure of a potential experiment containing fissile material are bounded by the MHA as defined by PUR-1. In addition, discuss the basis for limiting the maximum allowable fissile content of fueled experiments and how the limit is to be controlled either by a TS or by other acceptable means.

10. NUREG-1537, Part 2, Section 13, suggests that the definition of the maximum hypothetical accident (MHA) should be based on either a fuel plate or a fueled experiment failure, whichever leads to higher consequences. In the updated PUR-1 SAR, Sections 13.1.1 and 13.1.6, it is stated that the failure of a fueled experiment containing fissile material is the designated MHA. However, in the updated PUR-1 SAR, Section 13.2.1, an analysis is provided for the designated MHA that is the failure of a fuel plate. Explain how this is consistent, or correct the updated PUR-1 SAR, Section 13, with regard to the MHA.

11. NUREG-1537, Part 2, Section 13, defines the MHA as the failure of the cladding of one face of one fuel plate while suspended in air. PUR-1 SAR Section 13.2.1 defines the MHA as the failure of one face of one fuel plate submerged in the reactor pool resulting in a potentially nonconservative amount of radioactive iodine release into the reactor air volume. Discuss whether the assumption of radioactive fission product release into the pool water is a conservative assumption. Discuss whether the assumption bounds a failure of the fuel plate cladding while the fuel element is suspended in air, releasing fission products directly into the reactor air volume.

12. Provide an MHA safety analysis that explains all analyses, assumptions and conclusions at the requested licensed power level for the maximum potential estimate of the total radioactive fission product release after the failure of one side of one fuel plate. Discuss methodological assumptions associated with the following analytical steps:

    (a) Derivation of fission product atmospheric dispersion factor, χ/Q using either the methodology suggested in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, issued February 1983, or another equivalent method.

    (b) Dose conversion calculations using the Environmental Protection Agency's Federal Guidance Report (FGR)-11 and FGR-12 dose conversion coefficients or another equivalent methodology to account for inhalation/ingestion and submersion exposures.

13. Provide an analysis and discuss the potential maximum radiological dose estimate due to the MHA at the following suggested locations:

    (a) Facility worker – located inside the restricted area considering any evacuation procedure and potential residence time for staff exposed to fission product inhalation/ingestion and direct gamma ray radiation. The exhaust system operational status should be consistent with conservative assumptions.

    (b) Members of the public – located in adjacent, publicly accessible areas inside the reactor building (i.e., classrooms, hallways, adjacent rooms) potentially exposed to fission product inhalation and/or gamma ray radiation, taking into account any procedural process for evacuation, including the emergency plan. The exhaust system operational status should be consistent with conservative assumptions.

    (c) Members of the public – located outside the reactor building (maximally exposed location, nearest dormitories, offices, etc.) exposed to fission product inhalation/ingestion released from the reactor building and gamma ray radiation. The exhaust system operational status should be consistent with conservative assumptions.

14. 10 CFR Part 20, Standards for Protection Against Radiation," provides the regulatory framework and NUREG-1537, Part 1, Section 13.1.3 provides the guidance for licensees to systematically analyze and discuss credible accidents in each accident category. Section 13.1.3 of the updated PUR-1 SAR, describes the loss of coolant accident (LOCA) scenario. The updated PUR-1 SAR does not include an estimate for radiation levels in the reactor floor and the roof areas, due to the unshielded reactor core, after a postulated large LOCA event. The SAR should provide the consequent maximum dose rates at various locations on the reactor floor and outside on the reactor building roof. In accordance with 10 CFR Part 20, provide the accumulated doses to reactor building occupants and the maximally exposed member of the public, considering evacuation procedure and potential residence time for staff. In addition, provide an estimate when facility staff may enter the reactor building to start recovery operations.